FES Joule Milestone 2009
Second Quarter Report
March 31st, 2009

Annual Target: Conduct experiments on major fusion facilities to develop understanding of particle control and hydrogenic fuel retention in tokamaks. In FY09, FES will identify the fundamental processes governing particle balance by systematically investigating a combination of divertor geometries, particle exhaust capabilities, and wall materials. Alcator C-mod operates with high-Z metal walls, NSTX is pursuing the use of lithium surfaces in the divertor, and DIII-D continues operating with all graphite walls. Edge diagnostics measuring the heat and particle flux to walls and divertor surfaces, coupled with plasma profile data and material surface analysis, will provide input for validating simulation codes. The results achieved will be used to improve extrapolations to planned ITER operation.

Quarter 2 Milestone

Initial planned experiments will have been carried out on at least one of the three facilities.

Completion of 2nd Quarter Milestone

The 2nd quarter milestone has been completed by carrying out initial experiments on the facilities. This quarter saw significant activity towards overall research planning at the facilities, including coordinated research towards the 2009 Joule milestone. Research progress and activities are organized by facility below. It is important to note that substantial efforts are going into collaborative work, in particular the development of particle measurement techniques that can be used on the different facilities.

NSTX

The NSTX team held its annual Research Forum 8-10 December 2008 and retention experiments for the FY09 Joule milestone were given the highest priority. The following experimental proposals support the FY09 Joule milestone and were provisionally allocated run time in the Boundary Physics Topical Science Group and Lithium Research Thrust:

- FY2009 Retention and pumping milestone – 2-3 day
- Pedestal fueling comparison with SGI and gas - 0.5 day
- Dust transport and modeling – 0.5 day
- Experiments with lithium powder – 1.5 days
- Pre-LLD discharge characterization – 1 day

Initial experiments were carried out on NSTX. Additional FY09 NSTX experiments are scheduled to begin on April 3, 2009. A discharge was developed with a plasma current rampdown tailored to avoid reconnection events or disruptions that could release gas from the wall. The vessel pressure rise was measured after ohmic and Radio Frequency heated discharges with all pumping valves closed. The vessel pressure rise and evolutions was compared to a shot when gas-only was injected into the device, in order to isolate the effect of the plasma. At the conclusion of the experiments the pumping valves were left closed for ~ 24 h to monitor long term outgassing of fuel volatiles from the wall surfaces. One experiment was performed prior to lithium conditioning (a coating of lithium applied to the graphite wall) and one with lithium conditioning.

The vessel pressure immediately following a plasma discharge was rather low (Fig. 1) showing high deuterium retention of 97% – 100% in both pre-lithium and with-lithium cases immediately following the end of the plasma shot. Outgassing from the walls caused the pressure to rise after a discharge. This rise in pressure continued, however the rate of pressure rise decreased until it became comparable to the baseline vessel leak rate after about 12 hours (a finite leak rate is unavoidable in a tokamak vacuum system). This indicates that after 12 hours there is no more effective recovery of the hydrogenic fuel from the vessel walls. After 24 h the global retention (i.e. gas remaining in the vessel normalized to gas injected in a shot) had dropped to 50% in the pre-lithium case (Fig. 2) and 60% in the with-lithium case.

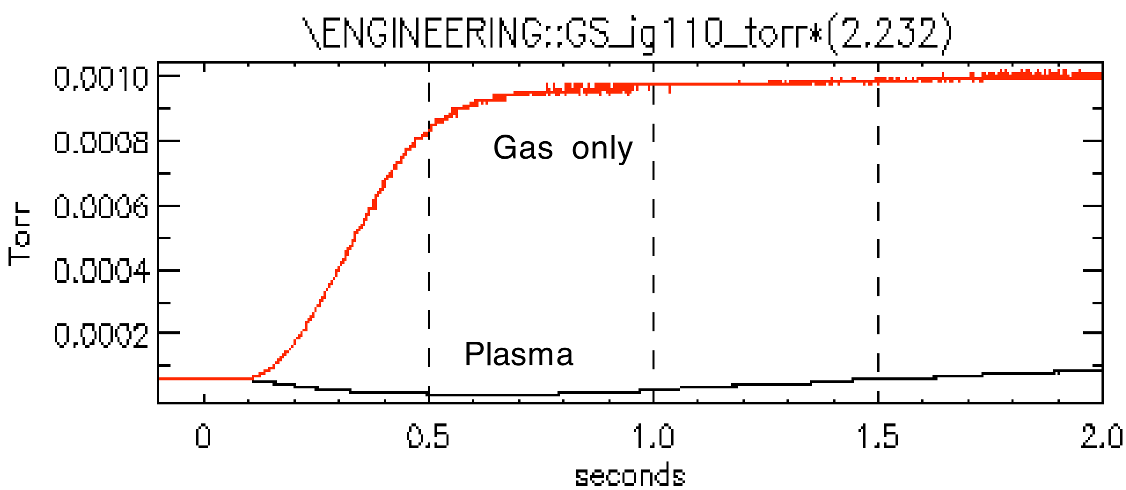

Fig. 1 Preliminary comparison of the NSTX gas-only pressure rise in torr with the pressure rise with a plasma. In both cases all the vessel pumping valves were closed. The initial retention is 97% - 100%. The plasma duration is ~0.6 seconds.

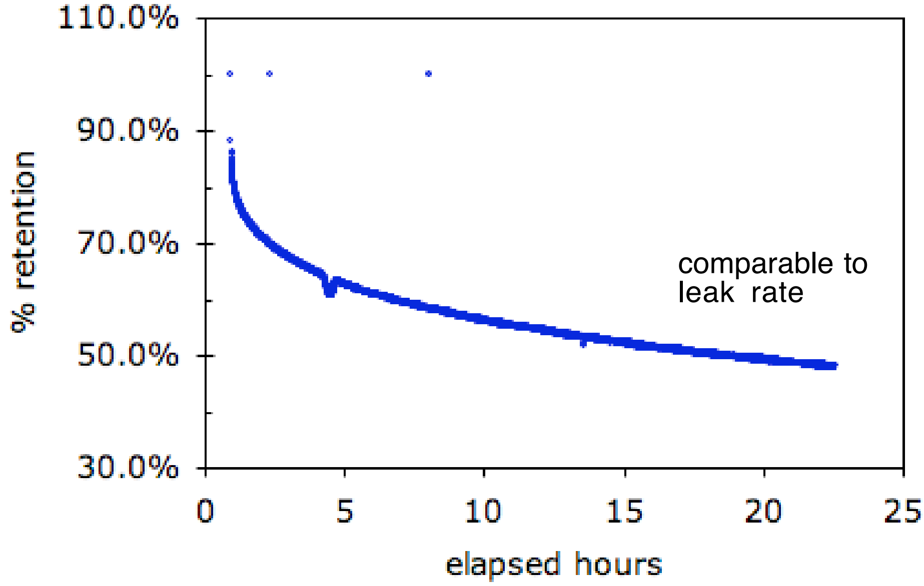

Fig. 2 Preliminary data showing NSTX long term retention decreasing as the plasma facing components outgas after a day's campaign. The data above was before the start of lithium conditioning. With lithium conditioning the behavior was similar but asymptoted to 60%.

DIII-D

The DIII-D Research Opportunities Forum (ROF) was held December 16-18, 2008 and experiments for the Joule Milestone were submitted in the new physics area of Plasma Boundary Interfaces with Tony Leonard as the physics manager of this group. Within this area, experiments were proposed to a newly formed High Priority Working Group: Hydrogen Retention. The purpose of this new group was described by a memo from DIII-D management. "Tritium Retention in a major driver for the choice of Plasma Facing Components in ITER, and DIII-D is uniquely positioned to evaluate plasma-wall interactions with all-carbon PFCs. Topics of interest include quantifying carbon migration pathways, developing methods for mitigating carbon co-deposits, and testing techniques for removal of hydrogen species from co-deposits (e.g. air bake). Activity for this topic directly supports the OFES FY09 Joint Facilities JOULE milestone and DIII-D 2009 Milestone 168, and will be organized within the Plasma Boundary Interfaces (PBI) group." In addition, two new Physics Area WGs also support the Joule Milestone: SOL Main Ion and Impurity Flows, and General Plasma Boundary Issues (including plasma materials interactions and DiMES experiments). After the submission of proposals to the ROF, several discussion sessions were held with the experimental staff in January, 2009 to develop a prioritized experimental plan for the Plasma Boundary Interfaces area, and this was presented to the Research Council in January, 2009. In the current run plan, the PBI Area received an allocation of five run days, and four of these contribute directly to the Joule milestone.

The list of relevant experiments and their scheduled run dates are as follows.

Static Particle Balance with C-MOD	2008 run campaign – 1 day
^{13}C experiment – ITER secondary divertor	End of 2008 campaign
65-01 Secondary Divertor Characterization	½ day completed 2/27/09
61-02 Static Particle Balance in ELMing H-mode & ECH	Scheduled 5/22/09
61-01 ^{13}C Injection followed by air bake	At end of run campaign
61-1 Diamond PFC Exposure (Mast Samples)	Scheduled 4/13/09 (1/2 day)
65-03 DiMES heated sample	Scheduled 4/17/09

Experiment 61-02 is being developed by Ali Mahdavi with the C-MOD experimental group, and is specifically designed to look at particle balance issues for this milestone. 61-01 is a test of tritium removal techniques in the DIII-D tokamak, and has four parts: a) First deposit a layer of ^{13}C by injection into 10-15 identical plasma shots, b) Remove a "before" set of tiles for analysis, c) Air Bake DIII-D at 10 Torr, 350 C for 2 hours, and d) Remove a second "after" set of tiles for analysis, and e) Re-establish plasma conditions. We are using the experimentally demonstrated highly-toroidally-symmetric injection and deposition of ^{13}C to compare ^{13}C co-deposits on the tiles before and after the air bake. Experiments 61-1 and 65-03 both involve the DiMES insertable probe; 61-1 is examining a effectiveness of a carbon diamond sample, and 65-3 is examining the effect of tile temperature on deposition efficiency. Previous results have shown that co-deposition can be dramatically reduced by heating a sample such as a mirror.

DIII-D has carried out part of its planned experiments. Towards meeting the Quarterly target, ½ day of experiments 65-01 was completed on 2/27/09. The goal of this experiment was to characterize the plasma at the divertor "region" opposite the active divertor in a single null (SN) plasma; this simulates the plasma conditions of ITER which is a SN plasma but has a weak divertor at the top of the machine (x-point still in the vacuum vessel). In the DIII-D experiment, the secondary divertor was located at the bottom of the machine (ITER turned upside down) to take advantage of the enhanced divertor diagnostic set in the lower divertor. These data will provide information for ITER on the plasma conditions in this region, and shown in Fig. 1 is an IRTV image from the new LLNL high-speed IRTV system (LIRT) showing ELMs in the lower (secondary) divertor. These data will also help us interpret the ^{13}C experiment that was completed last year. In this experiment carried out on 8/12/08, the plasma was run in an upper single null plasma shape (upside down ITER) and ^{13}C was injected in the toroidally symmetric plenum at the bottom of the machine. At the end of the campaign, several tiles were removed and sent to Sandia Labs, Albuquerque for analysis of the distribution of ^{13}C. We have previous ^{13}C experiments that show that most of the ^{13}C is deposited at the inner divertor in a lower single null (ITER-like) ELMing H-mode plasma when this secondary x-point was not in the vacuum vessel.

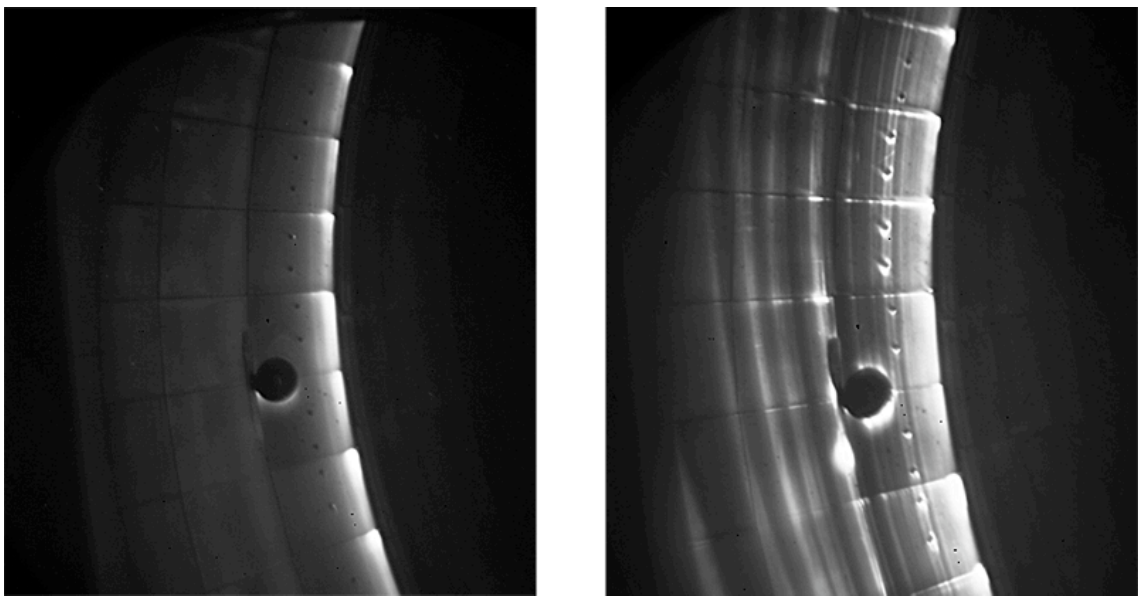

Fig. 3 Preliminary data from DIII-D with fast infrared imaging showing how Edge Localized Modes heat divertor targets. Accurate thermal diagnosis of the wall materials is an important element towards understanding particle balance. This data was obtained during the ITER secondary x-point experiments carried out during the quarter.

In support of 61-01, the air bake experiment, we have a set of experiments being carried out at University of Toronto and also more recently at the DIII-D Air Bake test stand supplied by LLNL. The goal of these tests is to determine "collateral" effects caused by air baking, i.e. damage to systems or degradation of diagnostics. A large number of tests were carried out at Toronto, and a Univ. Toronto graduate student reported these at the APS meeting in November.

Over 15 tests, each requiring one day, have been carried out on the DIII-D test stand. Much of this work has focused on the effects of air baking on copper components in DIII-D. Recently, we have also done the first "mixed" tests, e.g. a DIII-D carbon tile that has thick co-deposits and a diagnostic mirror. In these tests, co-deposits are removed by the air bake and we are testing if they are redeposited on the mirror. So far, any changes in reflectivity of the mirror as a function of wavelength have been barely detectable. A design review for the DIII-D air bake is schedule for the next quarter. In addition, we are having ongoing discussions with the JT-60 team about performing an air bake on this machine, as it is being de-commissioned in the fall of 2009.

DIII-D has performed experiments with both dynamic (during the shot) and static (pumps closed) particle balance. Shown in Fig. 4 is a dynamic particle balance using the analysis technique developed previously by R. Maingi, et al. (Partbal code). This is from a manuscript submitted to Nuclear Fusion Letters by E. A Unterberg, et al.

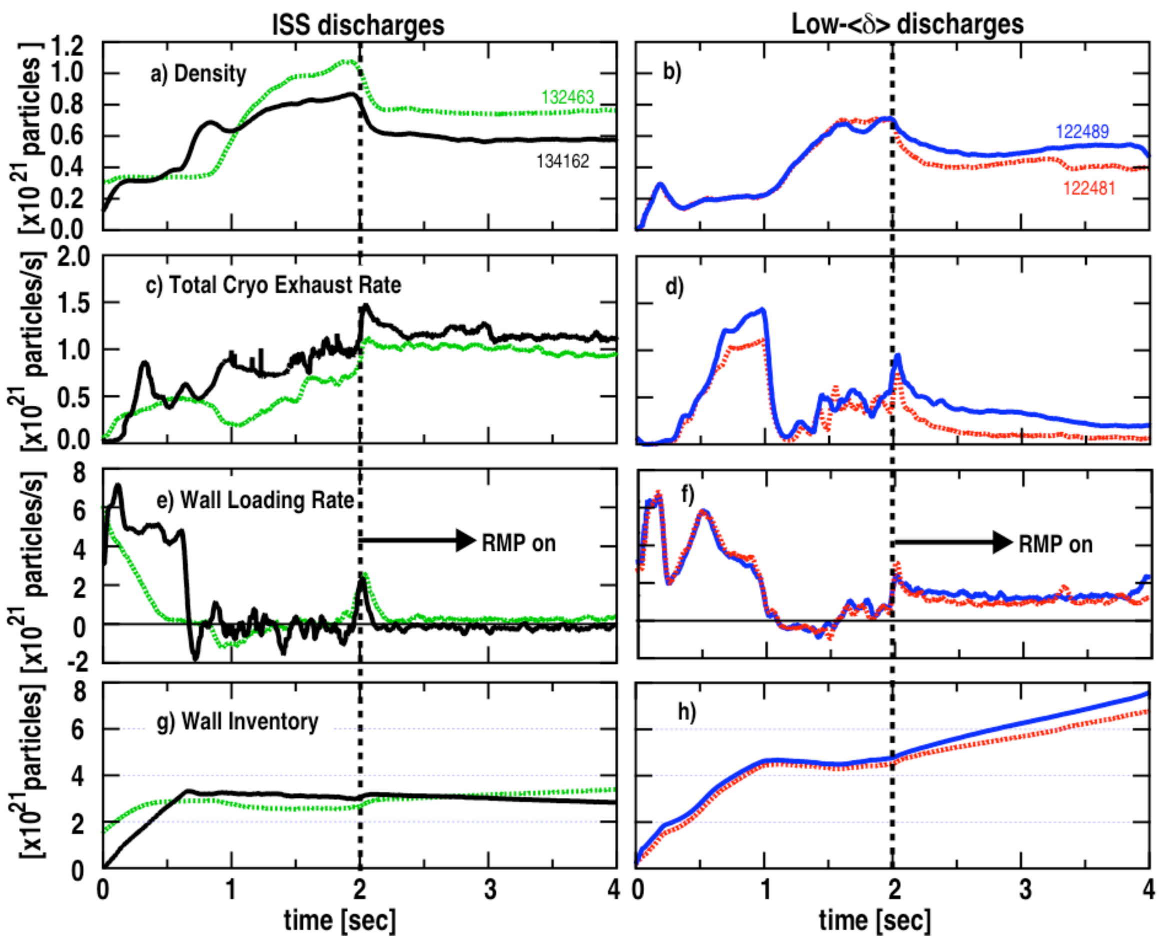

Fig. 4. From E. A Unterberg, et al., "Demonstration of Particle Exhaust Control During Resonant Magnetic Perturbtations in DIII-D, submitted to Nucl. Fus. Lett. Note the wall loading rate is large during the plasma startup (0-.5s), but is small during the time period after 3s, particularly for the discharge on the left with ELM control.

A joint experiment aimed at static particle balance was carried out with C-MOD last fall, and was the topic of an APS poster (P. West, et al., Bull. APS, Nov. 2008). The goal of this experiment was to obtain data with all of the pumping sources closed off – these experiments were pioneered by the C-MOD group. Fig. 5 shows the results of monitoring the vessel pressure after a L-mode discharge; note the time scales are different than the NSTX and C-MOD cases. As in NSTX and C-Mod the pressure rise asymptotes to a level consistent with the global leak rate, and in the DIII-D case, it is estimated that the wall can uptake 80% of the particles during this ohmic shot. This is consistent with Figure 4 above, where there is a large uptake during the ohmic and L-mode startup of the shot (0-.5s), followed by a period where there is little uptake. This can also depend on whether cryopumping is being used. DIII-D has 1 run day in the upcoming campaign to do these static particle balances in ELMing H-mode, and with ECH heating replacing neutral beam heating.

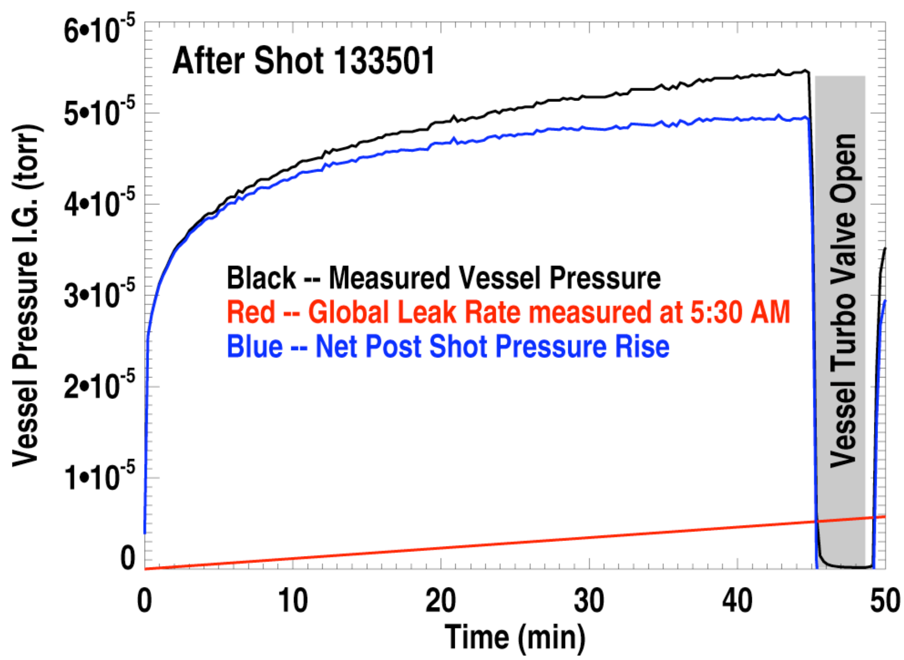

Fig. 5 Preliminary data from DIII-D experiment with no cyopumping or external pumps after L-mode discharge 133501. (P. West, et al., APS 2008) These data indicate that in the L-mode portion of the shot, there can be 80% uptake by the wall. These experiments will be repeated in ELMing H-mode during the upcoming campaign.

Alcator C-Mod

Alcator C-Mod is holding its 2009 research "Ideas Forum" from April 6-8. This is a preliminary list (in no particular order) of experimental ideas proposed towards the science goals of the 2009 Joule milestone. The prioritization and selection of C-Mod experiments will be carried out over the next month.

1. Fueling, particle transport and the density limit

2. The effect of fusion-reactor He fractions on D retention in Mo

3. Static vs. dynamic particle balance: Understanding the effects of confinement mode and divertor pumping

4. Helium particle control: Tau He* measurements with a fluid neutrals regime divertor

5. Role of particle transport and neutral penetration on plasma density for L-mode and H-mode confinement regimes without core fuelling.

Significant preparation work has been carried out on Alcator to prepare for the 2009 Joule Milestone, including plasma experiments on C-Mod. As mentioned above, the C-Mod group pioneered the development of static gas balance to improve particle accounting accuracy. Fig. 6 shows how the technique is deployed on a standard C-Mod shot [B. Lipschultz, et al. Nucl. Fusion 2009]. The active pumping is removed until several 100 seconds after the plasma ends, and at which time the vessel gas pressure has reached a steady value. The technique removes uncertainties associated with dynamic particle balance because it does not depend on knowing pumping speeds. However it

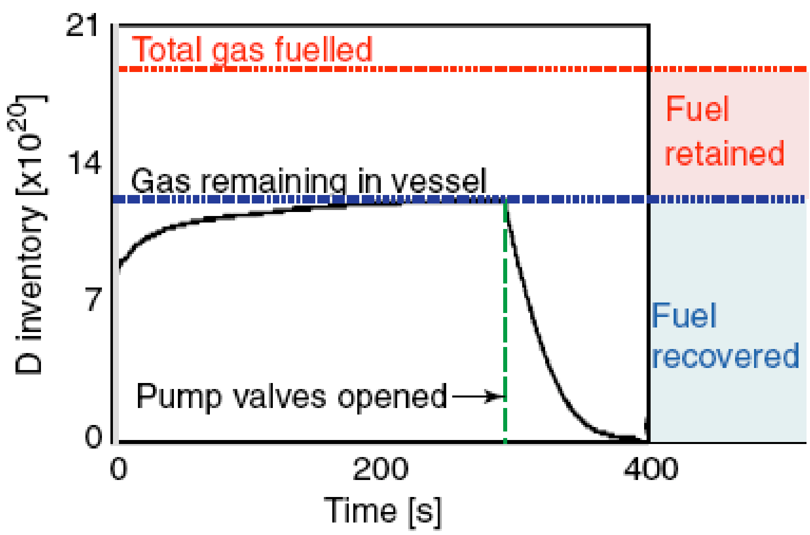

Fig. 6 Alcator C-Mod static particle balance experiment.

suffers from no time resolution during the shot, which dynamic particle balance provides. Proposed C-Mod experiments, as well as those on DIII-D and NSTX, will examine differences in these measurement techniques.

Another feature of the C-Mod experiments is that the pressure can also evolve on very long time-scales (e.g. 50 hours over a weekend of no plasma operation). This is qualitatively similar to the observations on NSTX (Fig. 2) and DIII-D (Fig. 5) that the rate of vessel gas pressure rise will approach the global leak rate of the vacuum system, indicating that long-term recovery of the retained fuel by the pumping system is not efficient. The results of particle fuel recovery from several weekends are shown in Fig. 7. Apparently the fuel recovery is independent of the number of plasma discharges prior to the weekend. Also, the quantity of recovered fuel is small; typical of the total retained D quantity from 3-4 non-disrupting C-Mod shots. Therefore the preliminary conclusion is that the fuel is not efficiently recovered from the walls by simply waiting for long periods of time; an observation that is important towards exploring possible retention mechanisms in NSTX, DIII-D and C-Mod.

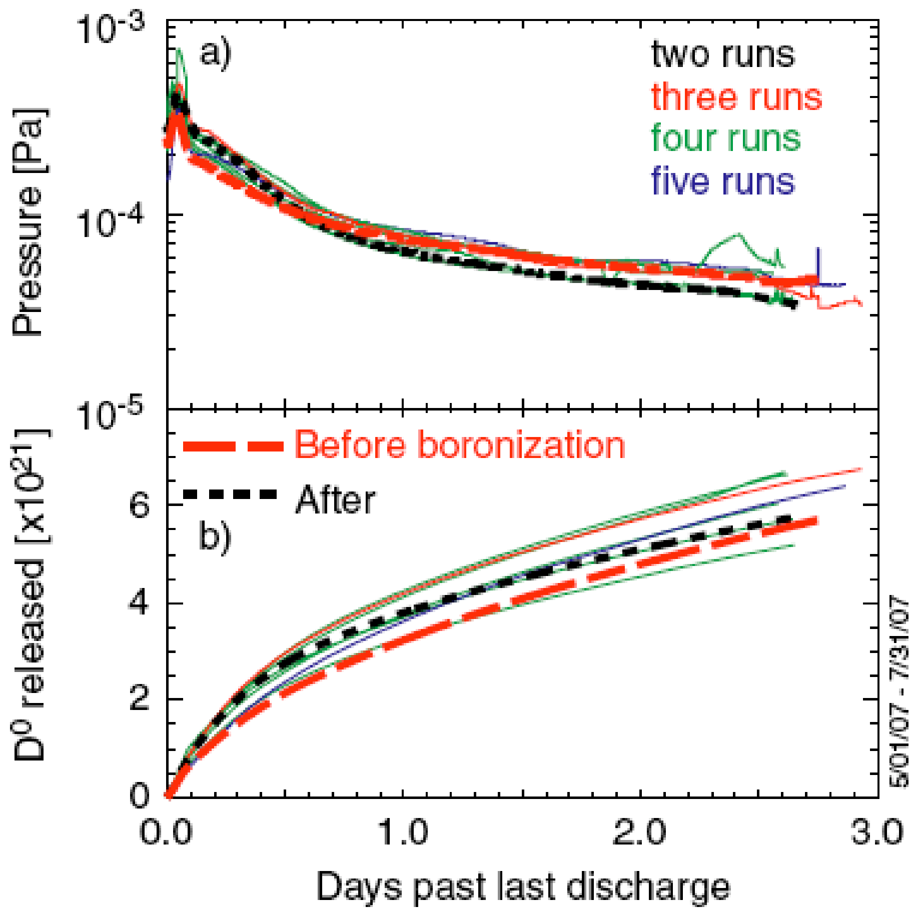

Fig. 7 Alcator C-Mod (top) Pressure evolution over weekend (pumped vessel). ((bottom) Recovered D fuel atoms is independent of the number of plasma discharges and/or vessel boronization.